June 16, 2000

Mr. William O'Connor, Jr.
Vice President
Nuclear Generation
The Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI INSPECTION REPORT 50-341/2000003(DRP)

Dear Mr. O'Connor:

On May 19, 2000, the NRC completed an inspection at your Fermi 2 reactor facility. The results were discussed with you and other members of your staff. The enclosed report presents the results of that inspection.

The inspection was an examination of activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel. Specifically, this inspection focused on resident inspection activities.

Based on the results of this inspection, the NRC has identified four issues that were evaluated under the risk significance determination process as having very low safety significance (GREEN). These issues have been entered into your corrective action program. The NRC has also determined that violations are associated with these issues. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VI.A of the Enforcement Policy. The issues involved two separate occasions where the dedicated shutdown cooling system was inadvertently stopped during the refueling outage, and another instance where the wrong oil was added to the emergency diesel generator bearings that rendered the diesel inoperable. Also, your staff discovered that a leaking containment isolation valve, for which enforcement discretion was granted in September of 1999, was caused by insufficient maintenance instructions for reassembling the valve during a previous outage. These issues are listed in the summary of findings and are discussed in the report. If you contest a violation or the significance of these NCVs you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-001, with copies to the Regional Administrator, Region III, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001, and the NRC Resident Inspector at Fermi.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the ADAMS Public Library component on the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (The Public Electronic Reading Room).

Sincerely,

/RA/

Mark A. Ring, Chief
Reactor Projects Branch 1

Docket No. 50-341
License No. NPF-43

Enclosure: Inspection Report 50-341/2000003(DRP)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	DPR-43
Report No:	50-341/2000003(DRP)
Licensee:	Detroit Edison Company
Facility:	Enrico Fermi, Unit 2
Location:	6400 N. Dixie Hwy. Newport, MI 48166
Dates:	April 2 through May 19, 2000
Inspectors:	S. Campbell, Senior Resident Inspector J. Larizza, Resident Inspector
Approved by:	Mark Ring, Chief Reactor Projects Branch 1 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

Enrico Fermi, Unit 2
NRC Inspection Report 50-341/2000003(DRP)

The report covers a 7-week period of resident inspection.

Cornerstone: Mitigating Systems

- GREEN. On April 22, Division 1 shutdown cooling was inadvertently stopped for 48 minutes. The cause of this event was that the core spray system safety tagging record specifying which fuse was required to be removed was not clear. Consequently, the wrong fuse was pulled causing an inadvertent engineered safety feature actuation, the closure of the Division 1 and 2 shutdown cooling inboard isolation valve, and a trip of the Division 1 residual heat removal pump A.

  The interruption of the shutdown cooling flow was evaluated using the Significance Determination Process. The event was found to be of very low risk because reactor vessel water level was above the reactor vessel flange (635 inches) and the time to boil was greater than 2 hours. The failure to develop an adequate safety tagging record was considered to be a non-cited violation of Technical Specification 5.4.1 per the NRC Revised Enforcement Policy. (Section 1R13)

- GREEN. On April 17, the Division 2 shutdown cooling system was inadvertently stopped for 21 minutes because an operator failed to recognize that only one of two power supplies provided power to logic trip unit B31N611B. One power supply was previously de-energized for planned modifications and the operator de-energized the second power supply. Safety Tagging Record 00-0501, required verifying two power sources to the trip unit. The failure to follow Safety Tagging Record 00-0501 was considered a non-cited violation of Technical Specification 5.4.1 per the NRC Enforcement Policy.

  The interruption of the shutdown cooling flow was evaluated using the Significance Determination Process. The event was found to be of very low risk because reactor vessel water level was above the reactor vessel flange (635 inches) and time to boil was greater than 2 hours. (Section 1R20)

- GREEN. On March 3, 2000, while conducting 18-month Preventive Maintenance Task W836000100, maintenance personnel added the incorrect oil (oil with too low viscosity) to the inboard and outboard bearings for Emergency Diesel Generator (EDG) 11. Low oil viscosity may cause bearing spalling (metallic deterioration of the bearing race) during EDG operation. An analyses could not guarantee machine operation for the 7-day requirement as specified in Updated Final Safety Analysis Report Chapter 8.3, Document R30-00, “Emergency Diesel Generators.” The past operability determination on Condition Assessment Resolution Document 00-15051 considered the EDG inoperable until April 12, when the correct oil was added to the bearings. The failure to add the correct oil in the inboard and outboard bearings for EDG 11 was considered a non-cited violation of Technical Specification 5.4.1 per the NRC Enforcement Policy.

The inoperable diesel was evaluated using the Significance Determination Process for the dates between March 3, 2000, and April 1, 2000, when the plant was operating at 97 percent power. Also, the inoperable EDG was evaluated using the Significance Determination Process between April 2 and April 12, 2000, which was the period when the unit was shutdown until the bearing oil was changed. In both evaluations, the risk significance was considered very low because the remaining three EDGs (12, 13, and 14) were available. (Section 1R22)

- GREEN. Drywell purge valve T4803F601 leaked excessively during local leak rate testing on September 22, 1999. Enforcement discretion was granted to allow non-compliance with Technical Specification 3.6.8.1. A follow-up investigation performed during the April 2000 refueling outage determined that the valve limit switches were not set properly in a previous outage. As a result, the valve tended to travel past the seat and pull the o-ring off the seat during repetitive valve stroking.

  The issue was considered to have very low risk significance because outboard valves T4800F407 and T4800F408 did not leak during the local leak rate testing conducted on September 22, 1999. However, the failure to provide adequate documentation to reassemble valve T4803F601 was considered a non-cited violation of Technical Specification 5.4.1 per the NRC Enforcement Policy. (Section 1R22)

Human Performance

- NO COLOR. The inspectors identified that errors in developing and implementing safety tagging records caused two occurrences of an intermittent loss in shutdown cooling flow (Sections 1R13 and 1R20). Errors also led to the incorrect performance of a safety-related surveillance test during the refueling outage (Section 1R04), and the addition of the incorrect oil to emergency diesel generator bearings (Section 1R22). While the risk of the individual events was determined to be very low (GREEN), human performance errors during operations and maintenance activities were evident. (Section 40A4)

Report Details

Plant Status

During the period, Unit 2 was shut down for the seventh refueling outage. On May 16, 2000, at 9:25 p.m.,operators began to withdraw control rods and the reactor went critical at 3:43 a.m., on May 17. After problems were resolved with the north reactor feed pump, the plant was at 2 percent reactor power at the end of the inspection period.

1. REACTOR SAFETY

1R04 Equipment Alignments

.1 Partial Walk Down of the Division 1 Emergency Diesel Generators (EDGs)

a. Inspection Scope (71111-04)

On April 21, 2000, the inspectors used Drawing 6M721-5734, "EDG System Functional Operating Sketch," Revision AB, to verify proper valve alignment for standby condition of the Division 2 EDGs.

b. Issues and Findings

There were no observations or findings associated with this inspection.

.2 Core Spray Pumps Improperly Aligned for Test

a. Inspection Scope (71111-04)

The inspectors reviewed the licensee's investigation results associated with CARD 00-16237. Documented on the CARD was the failure of the Division 2 core spray system (CSS) pumps to start while conducting Section 3.5.2 of Procedure 24.203.03, "Division 2 CSS Simulated Automatic Actuation Test." The inspectors reviewed Technical Specifications, CARD 00-16237, Procedure 24.203.03, and the outage risk plan.

b. Issues and Findings

There were no observations or findings identified.

1R05 Fire Protection

.1 Review of Fire Protection Areas and CARDs

a. Inspection Scope (71111-05)

The inspectors toured the reactor building refueling area, reactor recirculation motor generator area, standby liquid control system area, hydraulic control unit area, high pressure coolant injection pump area, and the RHR building complex that included the EDG rooms to ensure that any transient combustible materials and ignition sources were adequately controlled. The inspectors reviewed CARDs 00-13447, 00-14735, 00-12865, 00-16061, 00-10850, and 00-12864 to determine if fire protection related problems were adequately addressed in the licensee's corrective action program.

b. Issues and Findings

There were no observations or findings associated with this inspection.

1R07 Heat Sink Performance

.1 RHR Heat Exchanger Performance Test

a. Inspection Scope (71111-07)

The inspectors observed a portion of the performance of Procedure 47.205.01. "RHR Division 1 (North) Heat Exchanger Performance Test," and reviewed data collected during the test. The inspectors reviewed 1996 and 1998 data for previous Division 1 heat exchanger tests and examined the performance trending. The inspectors also reviewed Electric Power Research Institute 107397, "Service Water Heat Exchanger Testing Guidelines" and NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

b. Issues and Findings

There were no observations or findings associated with this inspection.

1R12 Maintenance Rule Implementation

.1 Maintenance Rule Implementation for Division 1 and 2 EDGs

a. Inspection Scope (71111-12)

The inspectors reviewed historical CARDs for Division 1 and 2 EDGs, Nuclear Management and Resources Council 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and the licensee's maintenance rule program manual to determine whether the licensee had properly classified the EDG related activities per the maintenance rule program as required by 10 CFR 50.65.

b. Issues and Findings

There were no observations or findings associated with this inspection.

1R13 Maintenance Risk and Emergent Work

.1 Loss of Division 1 Shutdown Cooling

a. Inspection Scope (71111-13)

The inspectors reviewed Condition Assessment Resolution Document (CARD) 00-15146, Safety Tagging Record (STR) 00-3348, and Drawing 6I261-2045-57 to assess the adequacy of controls on risk-significant maintenance that resulted in the loss of Division 1 shutdown cooling.

b. Issues and Findings

Shutdown cooling was inadvertently isolated during maintenance work due to removal of the wrong fuse.

On April 22, 2000, Work Request 000Z000642 was initiated to repair the leaking primary containment radiation monitoring system outlet valve T50F451. Safety Tagging Record 00-3348 required de-energizing power from the control and auxiliary panel inboard relay cabinet for the Division 1 instrument rack by removing a fuse that powered T50F451. During the tagout, Division 1 residual heat removal (RHR) pump A was operating in the shutdown cooling mode. A second verifier was sent with the operator to provide independent verification for pulling the correct fuse.

While hanging the tag to deenergize T50F451, the operators were required to remove fuse A71B-F901 at position AA F18 in panel H11P622. Instead operators removed fuse A71-F62 at position AA F-12. Removal of this fuse de-energized the circuit for the Group 4 RHR shutdown cooling and head spray isolation logic. This initiated automatic closure of RHR Division 1 and 2 shutdown cooling inboard containment isolation valve E1150F009, causing a trip of the Division 1 RHR pump A.

Upon the loss of the shutdown cooling pump, operators entered Abnormal Operating Procedure 20.205.01, “Loss of Shutdown Cooling.” Operators restored shutdown cooling within approximately 48 minutes. The licensee documented the event on CARD 00-15146. Also, the licensee initiated Licensee Event Report (LER) 00-008 per 10 CFR 50.73(a)(2)(iv), for the inadvertent engineered safety feature actuation. During the investigation, the licensee determined that although sufficient information existed in the Safety Tagging Record (STR), it was not written clearly enough to ensure the correct fuse was removed. This was known to the nuclear supervising operator who approved the STR. However, since the individual who developed the STR was the same operator installing the tags, the nuclear supervising operator accepted the STR without clarification. Additionally, the independent verifier did not review drawings to confirm that the STR was correct.

The interruption of shutdown cooling was evaluated using the Significance Determination Process. The plant risk from losing shutdown cooling was very low since reactor water temperature was below boiling temperature of 212 degrees Fahrenheit, water level was 635 inches (above the reactor vessel flange), and the time to boil was greater than two hours.

Technical Specification 5.4.1. requires that written procedures shall be implemented for procedures recommended in Regulatory Guide 1.33. Revision 2, Appendix A, February 1978. Safety Tagging Record 00-3348 fits the guidance for a recommended procedure.

Failure to implement STR 00-3348 due to unclear tagging nomenclature was a violation of Technical Specification 5.4.1. However, this violation is considered a **non-cited violation (NCV 50-341/2000003-02)** consistent with the NRC Revised Enforcement Policy. This violation is in the licensee's corrective action program as CARD 00-15146.

1R15 Operability Evaluations

.1 Unexpected Pressure Transient in the Emergency Equipment Cooling Water System (EECW)

a. Inspection Scope (71111-15)

The inspectors reviewed CARD 00-15085 that documented an unexpected pressure transient in the EECW system while starting the emergency equipment service water (EESW) pumps to ensure that the operability evaluation was thorough and complete. The inspectors reviewed the following:

- CARD 00-15085,
- Technical Specifications,
- the accompanying engineering functional analysis,
- design basis information,
- vendor information,
- Operations Department Instruction ODI-50, "Operator Required Reading,"
- temporary change notices for Procedures 24.208.02, "Division 1 EESW Pump and Valve Operability Test," 24.207.06," and, "EECW/EESW Actuation Functional Test -Division 1,"

b. Issues and Findings

There were no observations or findings associated with this inspection.

.2 Removal of Several Open CARDs from the Resolution Action Items List

a. Inspection Scope (71111-15)

On May 4, 2000, the inspectors noted that, over one shift, several open CARDs had been removed from the Resolution Action Item List. The licensee had previously established a milestone that the CARDs be closed before removing the spent fuel pool gates. The open CARDs addressed calibration issues for measuring and test equipment that may have been used on safety-related equipment. The inspectors selected a sampling of the CARDs and reviewed the following to ensure the licensee's conclusions and operability evaluations in resolving the CARDs were appropriate:

- CARD 00-13178
- CARD 00-16285
- CARD 00-15897
- CARD 00-16283
- CARD 00-16169
- CARD 00-16279
- CARD 00-16171
- CARD 00-16292
- CARD 00-16282
- CARD 00-13177
- CARD 00-13176
- CARD 00-16284
- CARD 00-16293

b. Issues and Findings

There were no observations or findings associated with this inspection.

1R19 Post Maintenance Testing

.1 Post Maintenance Testing of Main Steam Isolation Valve C

a. Inspection Scope (71111-19)

The inspectors observed the performance of Procedure 43.401.300 "Local Leakage Rate Test Type 'C' - General for Main Steam Isolation Valve C," and reviewed applicable data collected during the test.

b. Issues and Findings

There were no observations or findings associated with this inspection.

1R20 Refueling and Outage

.1 Refueling Outage Inspections

a. Inspection Scope (71111-20)

The inspectors observed and evaluated outage activities including the outage plan, shutdown activities, cool down rates, outage configuration management, reactor coolant instrumentation, electrical power, decay heat removal system monitoring, spent fuel pool cooling system operation, inventory control, reactivity control, defense-in-depth and shutdown risk criteria.

The inspectors reviewed the following procedures:

- IWMG-8 “Refueling Outages”
- IWMG-10 “Outage Nuclear Safety”
- IWMG-13 “Operations Outage Management Organization”

The inspectors toured the primary containment (drywell) and the suppression pool (torus). In addition, the inspectors observed refueling activities that included the movement of fuel assemblies in the proper location in the core and observed foreign material exclusion practices for the refueling area.

b. Issues and Findings

There were no observations or findings associated with this inspection.

.2 Loss of Division 2 Shutdown Cooling (71111-20)

a. Inspection Scope

The inspectors reviewed Safety Tagging Records 00-3202 and 00-0501, CARD 00-11412, and Drawings 6I261-2225-10, 2105-12, 2201-16, 2201-05, 2205-05, 2205-06, and 2201-02 to determine the cause for the inadvertent loss of Division 2 shutdown cooling.

b. Issues and Findings

Failure to verify power from both inverters when implementing a tag-out led to the loss of shutdown cooling.

On April 6, 2000, the operators tagged-out inverter R31-K00R per STR 00--3202 to install the recirculation pump digital control system. Since inverter R31-K004 was tagged out, power to the trip unit was provided by inverter E21-K601B. Either inverter, R31-K004 or E21-K601B, provides power to the logic trip unit B31N611B. The trip unit, when de-energized, closes the Division 1 and 2 RHR shutdown cooling outboard isolating valve E1150-F008 and trips the pump.

On April 17, 2000, while RHR pump B was operating in the shutdown cooling mode, operators released STR 00-0501 to tag out the Division 2 core spray system. The tagout included removing fuses from inverter E21-K601B. Since inverter E21-K601B was the only power source to the trip unit for Divisions 1 and 2 RHR shutdown cooling outboard isolation valve, a note was added to STR 00-501 to ensure power to both inverters (by verifying white light indication on both inverters) to prevent de-energizing the trip unit. This was discussed in a pre-job brief before conducting STR 00-0501.

At 2:51 a.m., while removing the fuses from inverter E21-K601B, the operator missed seeing the note before pulling the fuses and pulled the fuses. The trip unit de-energized. Valve E1150-F008 closed and RHR pump B tripped causing a loss of shutdown cooling. The reactor steam dome pressure instrument loop was also de-energized which initiated the shutdown cooling system isolation on high steam dome pressure. Operators in the control room entered Abnormal Operating Procedure 20.205.01, “Loss of Shutdown Cooling.” An operator reinstalled the fuses and shutdown cooling was restored within 21 minutes. The licensee initiated CARD 00-11412 to document this event. Also, the licensee initiated LER 00-006 per 10 CFR 50.73(a)(2)(iv) for the inadvertent engineered safety feature actuation.

The interruption of shutdown cooling was evaluated using the Significance Determination Process. The plant risk from losing shutdown cooling was very low (GREEN) since reactor water temperature was below boiling temperature of 212°F, water level was 635 inches (above the reactor vessel flange), and time to boil was greater than 2 hours.

Technical Specification 5.4.1. requires that written procedures shall be implemented for procedures recommended in Regulatory Guide 1.33. Revision 2, Appendix A, February 1978. Safety Tagging Record 00-0501 fits the guidance for a recommended procedure. The failure to implement the note on STR 00-0501 is a TS 5.4.1 violation. However, this violation is considered a **non-cited violation (NCV 50-341/2000003-03)** consistent with the NRC Enforcement Policy. This violation is in the licensee’s corrective action program as CARD 00-11412.

1R22 Surveillance Testing

.1 Surveillances Observed and Reviewed

a. Inspection Scope (71111-22)

The inspectors observed the performance of the following tests:

- 23.800.07, “Reactor Coolant Natural Circulation and Decay Heat Removal”
- 24.139.03, “Standby Liquid Control Manual Initiation, Reactor Water Cleanup Isolation and Storage Tank Heater Operability Test”
- 24.307.04, “EDG 14 - Loss of Offsite Power and Emergency Core Cooling System Start with Loss of Offsite Power Test”

The inspectors reviewed applicable data collected during the tests. In addition, the inspectors reviewed the licensee’s temporary change notice associated with Procedure 24.307.04, the infrequently performed test or evolution review and approval request for Procedure 23.800.07 and the associated TS limiting conditions for operation.

b. Issues and Findings

The inspectors did not identify any findings with this inspection activity.

.2 Wrong Oil Added in EDG 11 Bearings

a. Inspection Scope (71111-22)

The inspectors observed the performance of surveillance activities on EDG 11, and reviewed CARD 00-15051 and the circumstances surrounding the discovery of wrong oil added to the inboard and outboard alternator bearings for EDG 11.

b. Issues and Findings

On March 3, 2000, while the plant was at 97 percent power, maintenance personnel performed preventive maintenance on EDG 11 per PM Task W836000100 and Procedure 4.307.001. The activity included draining, flushing, and filling oil in the EDG alternator bearings and adding oil in the generator governor. Two different brands of oil were available for the activity. Maintenance personnel failed to verify the correct label identifying the oil brand and added the wrong oil to the inboard and outboard alternator bearings. The incorrect oil was not within the normal oil viscosity range of between 200 to 240 centistokes. The maintenance was completed and, on March 4, the EDG tested satisfactorily per Procedure 24.307.014, Section 5.1, "EDG 11 Start and Load Test - Slow Start."

On April 11, 2000, while the plant was shutdown for the seventh refueling outage, the EDG was tested again per Procedure 24.307.014. The EDG alternator bearings were sampled and the licensee determined the viscosity at 100 degrees Fahrenheit was 64 and 99 centistokes for the inboard and outboard bearings, respectively. The licensee initiated CARD 00-15051 to document the condition. Mechanics added the correct oil to the alternator bearings on April 12. The alternator bearings for EDGs 12, 13, and 14 were sampled. The viscosity level was found acceptable for these EDGs.

Oil samples from EDG 11 were sent to three independent laboratories to determine the expected life of the bearings under operating conditions. At higher operating temperatures, viscosity decreases as the bearings heat up. Low viscosity levels may not be sufficient to protect the bearing from spalling (metallic breakdown of the bearing race) during engine operation. After analysis, none of the laboratories could guarantee engine performance per the 7-day requirement as specified in Updated Final Safety Analysis Report, Chapter 8.3, Document R30-00, "EDG," with low viscosity levels. The licensee performed a past operability determination for the incorrect oil in the bearings. The licensee determined that the EDG had been inoperable since March 3, 2000.

Technical Specification 5.4.1. requires that written procedures shall be implemented for procedures recommended in Regulatory Guide 1.33. Revision 2, Appendix A, February 1978. PM Task W836000100 and Procedure 4.307.001 fit the guidance for a recommended procedure. The failure to add the correct oil in EDG 11 bearings was considered to be a TS 5.4.1 violation. However, this violation is considered a **non-cited violation (NCV 50-341/2000003-04)** consistent with the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CARD 00-15051.

The inoperable diesel was evaluated using the Significance Determination Process for the dates between March 3, 2000, and April 1, 2000, when the plant was operating at 97 percent power. Also, the inoperable EDG was evaluated using the Significance Determination Process between April 2 and April 12, 2000, when the unit was shutdown until the bearing oil was changed. In both evaluations, the risk significance was very low (GREEN) because three EDGs (12, 13, and 14) were required and were available while the unit was at power and during hot shutdown conditions. During cold shutdown and refueling conditions, the Significance Determination Process indicated only one EDG was needed to ensure very low risk and at least one was available.

.3 Failed Emergency Diesel Generator Linear Reactors

a. Inspection Scope (71111-22)

The inspectors reviewed the following CARDs associated with linear reactor failures on the EDGs discovered during surveillance testing:

- CARD 98-18708
- CARD 99-17463
- CARD 99-18175
- CARD 00-17643
- CARD 00-15490
- CARD 00-15446
- CARD 00-14305

b. Issues and Findings

Within the past 2 years, the utility has experienced age-related failures of the linear reactors (transformers) on three of the four EDGs during diesel runs and tests. Each EDG has three linear reactors in the excitation circuit. The linear reactors provide base excitation voltage when the EDG is unloaded.

Linear Reactor 1 for EDG 11 failed a high potential test on May 9, 2000. A high potential test places a high voltage on the linear reactor winding to determine whether the winding is degraded. The remaining two EDG 11 linear reactors have experienced no failures, however, the licensee replaced linear reactors 2 and 3 with refurbished linear reactors from EDG 13. Linear reactors 2 and 1 on EDG 12 failed on October 8, 1998, and April 12, 2000, respectively, and were replaced. Linear reactor 3 on EDG 12 failed a high potential test on May 9, 2000, and the linear reactor was replaced. For EDG 14, linear reactor 1 failed a high potential test on May 8, 2000, and the linear reactor was replaced. Also, linear reactor 2 failed on October 21, 1999, and the linear reactor was replaced. No linear reactor failures occurred on EDG 13, however, the licensee replaced the EDG 13 excitation circuit with a circuit containing no linear reactors. The licensee planned to install excitation circuits that do not have linear reactors on the remaining EDGs in future planned outages.

Condition Resolution Assessment Documents were written for each condition. Linear reactors that experienced random failures (EDGs 12 and 14) were sent to a laboratory for failure analysis. The failures were caused by linear reactor winding degradation (i.e., degradation of the insulation winding).

The inspectors were concerned with the frequency, similar failure modes and the impact these failures may have on plant risk. This issue will be an **unresolved item (50-341/200003-05)** pending a review of the impact on plant risk from these failures.

.4 <u>(Closed) Unresolved Item 50-341/99014-01</u>: Determine cause of leak test failure of Penetration X-26. This item involved a failed leak rate test of nitrogen inerting drywell purge inlet supply valve T4803F601 that occurred on September 22, 1999. The valve leaked above the acceptance criteria of 14.87 standard cubic feet per hour (scfh) at 26.7 scfh. During the test, the licensee determined that the remaining valves (drywell purge inlet supply valve T4800F407 and drywell nitrogen supply outboard isolation valve 4800F408) in Penetration X-26 had leakage within the acceptance criteria. The licensee requested enforcement discretion not to comply with TS 3.6.8.1, Action B to shut down the plant provided that: 1) a blank flange be installed on the valve, 2) the valve be de-energized closed, 3) the remaining Penetration X-26 valves (T4800F407 and T4800F408) be verified closed every 31 days, and 4) Penetration X-26 be tested every 45 days. The licensee agreed to the compensatory measures and the NRC granted discretion. Subsequently, the compensatory measures were implemented.

During Refueling Outage 7, maintenance personnel disassembled the valve and found that the hard o-ring had separated from the valve disk. The separation was caused by not setting the motor operator valve limit switches to stop the valve stroke and prevent an over-travel of the valve disk within the valve seat. The over-travel pulled the hard o-ring from the valve disk.

The licensee conducted a root cause investigation and discovered that in Refueling Outage 6, maintenance personnel had replaced the soft o-ring with a hard o-ring under Work Request 000Z983555. The soft o-ring material was ductile enough to permit the valve disk to over-travel and allow o-ring deformation without valve disk separation. Therefore setting the limit switches was not required to be precise. However, the valve with the hard o-ring material required a precise limit switch setting to prevent over-travel and separation. The vendor provided instructions for the hard o-ring to set the limit switches properly. However, these instructions were not incorporated into plant maintenance procedures.

10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings requires that activities affecting quality shall be prescribed by documented instructions. The failure to provide adequate maintenance instructions for setting the containment isolation valve limit switches is a violation of 10 CFR Part 50, Appendix B. However, this violation is considered a **non-cited violation (NCV 50-341/2000003-06)** consistent with the NRC Enforcement policy. This violation is in the licensee’s corrective action program as CARDs 99-16855 and 00-15396.

The inspectors evaluated this issue using the Significance Determination Process and determined the risk significance to be very low (GREEN) since the outboard valves

(T4800F407 and T4800F408) did not leak during testing and would be sufficient to provide containment integrity. This item is closed.

1R23 Temporary Plant Modifications

.1 Temporary Repair on SCRAM Air Header for Hydraulic Control Unit 18-15

a. Inspection Scope (71111-23)

The inspectors reviewed Temporary Modification 98-0027, which installed a temporary patch on a brazed fitting for hydraulic control unit 18-15 to stop an air leak, and reviewed Procedure MES 12, "Performing Temporary Modifications."

b. Issues and Findings

There were no observations or findings associated with this inspection.

4. OTHER ACTIVITIES (OA)

4OA3 Event Follow-up (71153)

.1 (Closed) LER (50-341/00-008): Engineered Safety Feature Actuation: Invalid Automatic Closure of E1150F009 Resulting in Shutdown Cooling Interruption. This issue was addressed in Section 1R13.1 of this report.

.2 (Closed) LER (50-341/00-006): Engineered Safety Feature Actuation: Invalid Automatic Closure of E1150F008 Resulting in Shutdown Cooling Interruption. This issue was addressed in Section 1R20.2 of this report.

.3 (Closed) LER (50-341/00-009): Emergency Diesel Generator Inoperable Due to Incorrect Oil Installed in Generator Bearing. This issue was addressed in Section 1R22.2 of this report.

.4 (Closed) LERs (50-341/99-005-00 and 01): Containment Purge Isolation Valve Leak Rate Test Failure. This issue was addressed in Section 1R22.4 of this report.

4OA4 Cross-Cutting Issues

Human Performance Problems

a. Inspection Scope (71111-20)

The inspectors observed operations and maintenance activities that included operation of the shut down cooling system and conducting tests before and during the refueling outage.

b. Issues and Findings

The inspectors found that due to inattention to detail and poorly written safety tagging records, an unexpected loss of shutdown cooling occurred on Division 1 for 48 minutes (Section 1R013) and for 21 minutes on Division 2 (Section 1R20). At the time of the errors, over heating and boiling of the reactor coolant would not have occurred due to the large volume of water that existed during refueling. Therefore, these events were considered of very low risk significance (GREEN).

A meaninful maintenance error occurred during an 18-month preventive maintenance (PM) task approximately 1 month before the outage. Inattention to detail caused the wrong oil to be added to the alternator bearings for EDG 11 (Section 1R22). This error caused the EDG to be inoperable longer than the 7-day action statement allowed by TS 3.8.1.1. This issue was considered of very low risk significance (GREEN) because the minimum required EDGs remained operable.

4OA6 Management Meeting

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. O'Connor and other members of licensee management at the conclusion of the inspection on May 19, 2000. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. O'Connor, Vice President, Nuclear Operations
P. Fessler, Assistant Vice President, Nuclear Operations
R. Libra, Director, System Engineering
R. DeLong, Director, System Engineering
J. Moyers, Director, Nuclear Quality Assurance
S. Stasek, Manager, Nuclear Assessment
D. Cobb, Superintendent, Maintenance
K. Hlavaty, Superintendent, Operations
E. Kokosky, Superintendent, Radiation Protection
J. Davis, Outage Management
S. Booker, Work Control
P. Smith, Licensing
K. Howard, Plant Support, Engineering
J. Pendergast, Principal Engineer, Licensing
S. Peterman, Engineer, Operations
K. Harsley, Licensing
J. Flint, Licensing

NRC

M. Ring, Chief, Reactor Projects Branch 1
S. Campbell, Senior Resident Inspector
J. Larizza, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-341/2000003-01	NCV	Failure to Place Pumps in Automatic Before Testing the CSS Pumps
50-341/2000003-02	NCV	Unclear Tagging Nomenclature of STR 00-3348
50-341/2000003-03	NCV	Failure to Follow Note on STR 00-0501 and Fuses Pulled
50-341/2000003-04	NCV	Failure to Add Correct Oil in EDG 11 Bearings
50-341/2000003-05	URI	Review of the Impact on Plant Risk From Age Related Failures of Linear Reactors on Three of Four EDGs
50-341/2000003-06	NCV	Failure to Provide Adequate Maintenance Instructions for Setting Containment Isolation Valve Limit Switches

Closed

50-341/2000003-01	NCV	Failure to Place Pumps in Automatic Before Testing the CSS Pumps
50-341/2000003-02	NCV	Unclear Tagging Nomenclature of STR 00-3348
50-341/2000003-03	NCV	Failure to Follow Note on STR 00-0501 and Fuses Pulled
50-341/2000003-04	NCV	Failure to Add Correct Oil in EDG 11 Bearings
50-341/2000003-06	NCV	Failure to Provide Adequate Maintenance Instructions for Setting Containment Isolation Valve Limit Switches
50-341/00-008-00	LER	ESF Actuation: Invalid Automatic Closure of E1150F009 Resulting in Shutdown Cooling Interruption
50-341/00-006-00	LER	ESF Actuation: Invalid Automatic Closure of E1150F008 Resulting in Shutdown Cooling Interruption
50-341/00-009-00	LER	EDG Inoperable Due to Incorrect Oil Installed in Generator Bearing
50-341/99-005-00	LER	Containment Purge Isolation Valve Leak Rate Test Failure
50-341/99-005-01	LER	Containment Purge Isolation Valve Leak Rate Test Failure

Discussed

None

LIST OF BASELINE INSPECTIONS PERFORMED

The following inspectable-area procedures were used to perform inspections during the report period. Documented findings are contained in the body of the report.

Inspection Procedure Number	Inspection Procedure Title	Report Section
71111-04	Equipment Alignment	1R04
71111-05	Fire Protection	1R05
71111-07	Heat Sink Performance	1R07
71111-12	Maintenance Rule Implementation	1R12
71111-13	Maintenance Risk and Emergent Work	1R13
71111-15	Operability Evaluations	1R15
71111-19	Post Maintenance Testing	1R19
71111-20	Refueling and Outage Activities	1R20
71111-22	Surveillance Testing	1R22
71111-23	Temporary Plant Modifications	1R23
71150	Plant Status	
71153	Event Follow-up	4OA3

LIST OF ACRONYMS USED

CARD	Condition Assessment Resolution Document
CFR	Code of Federal Regulations
CSS	Core Spray System
EDG	Emergency Diesel Generator
EECW	Emergency Equipment Cooling Water
EESW	Emergency Equipment Service Water
LER	Licensee Event Report
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PM	Preventative Maintenance
RHR	Residual Heat Removal
STR	Safety Tagging Record
TS	Technical Specification